# AP1000®[1] Plant Operational Transient Analysis

LIU Lixin[1], ZHENG Limin[2]

Shanghai Nuclear Engineering Research and Design Institute (SNERDI)

29 Hong Cao Rd., Shanghai 200233, P.R. China

[1]liulixin@snerdi.com.cn; [2]zhenglimin@snerdi.com.cn

***Abstract***

According to advanced light water reactor (ALWR) Utility Requirement Document (URD), plant control systems should be designed to meet the specified design requirements for an advanced 1000 MWe class PWR nuclear power plant (NPP).

In this paper, as per the plant design requirements, AP1000 control systems were simulated, and plant thermal-hydraulic performance analysis have been performed with RELAP5 thermal-hydraulic computer code under five typical operational transients, i.e. (a) a step load increase from 15% Full Power (FP) to 25%FP, (b) a step load decrease from 100% FP to 90%FP, (c) +5%/min ramp load increase, (d) large load rejection to house load at full power, (e) normal reactor trip at full power. Based on a preliminary plant performance analysis, it indicates that plant design requirements are satisfied for AP1000 control systems.

***Keywords***

*PWR; Pressurizer; URD; Step/Ramp Load; Load Rejection; Operational Transient*

## Introduction

Main functions of Nuclear Steam Supply System (NSSS) control systems are to control plant operation with acceptable performance for specified normal operational transients and plant unanticipated events.

AP1000 NSSS control systems mainly consist of six different control systems, i.e. a) Reactor control system (RECS), b) Rapid power reduction (RPR), c) Steam dump control system (SDCS), d) Feedwater control system (FWCS), e) Pressurizer pressure control system (PPCS), f) Pressurizer level control system (PLCS). For each individual control system, different functionalities and different control modes are designed in consideration of plant operational conditions and severity of potential plant operational transients.

According to advanced light water reactor (ALWR) Utility Requirement Document (URD) [1], plant should be designed to meet the following design requirements under operational transients for advanced 1000 MWe class PWR nuclear power plant (NPP).

1) Pressurizer power-operated relief valves (PORV) are not required to mitigate overpressure transients (e.g. complete loss of load operational transient) and safety valves are not actuated by overpressure transients with all normal support systems available.

2) Plant should be capable of load rejection from 100 percent rated power without reactor trip or turbine trip and without lifting main steam safety valves, and be able to continue stable operation with minimal house load.

3) To maintain RCS pressure response resulting from operational transients (e.g. 10%FP step load increase or reduction transient) within an appropriate range such that the setpoint should not be reached for pressurizer high pressure/low pressure trip signal, high water level trip and safety injection actuation signals, and such that PORVs or primary safety valves should not be actuated. Credit may be taken for all normally available support systems (i.e. control systems available).

4) To maintain RCS inventory such that the minimum pressurizer level during operational transients is above the setpoint for low level reactor

[1] **AP1000®** is a trademark or registered trademark of Westinghouse Electric Company LLC, its Affiliates and/or its Subsidiaries in the United States of America and may be registered in other countries throughout the world.  Other names may be trademarks of their respective owners.

trip and safety injection. Credit may be taken for all normally available support systems.

5) To prevent pressurizer heater uncovery following a reactor trip and turbine trip. Credit may be taken for all normally available support systems (i.e. control systems available).

Based on a preliminary plant performance analysis, it indicates that plant design requirements are satisfied for AP1000 control systems.

## Methods and Assumptions

### *Initiating Events and Acceptance Criteria*

Based on the plant design requirements and related transient analysis [2][3], the initiating events and their acceptance criteria are summarized as follows.

1) Pressurizer low pressure trip signal should not be actuated under a step load increase transient from 15%FP to 25%FP.

2) Pressurizer high pressure trip signal should not be actuated under a step load reduction transient from 100%FP to 90%FP.

3) Reactor trip setpoint should not be reached, and pressurizer safety valve should not be actuated under +5%/min ramp load increase from 30% to 100%FP.

4) Reactor trip (pressurizer high pressure /high water level trips) or turbine trip should not be actuated under load rejection to house load at full power without lifting main steam safety valves, plant should be able to continue stable operation with minimal house load and pressurizer safety valves are not actuated by overpressure transients with all normal support systems available.

5) Pressurizer heater uncovery and low pressure safety injection actuation should be prevented following a reactor trip and turbine trip, with normal operation of control and makeup systems.

For the RCS cooldown transients, 10%FP step load increase transient and reactor trip transient were analyzed. A step load increase transient from 15%FP is selected because the reactor control system is initiated only when the reactor power is greater than 15%FP while the nominal pressurizer water level is relative lower. In addition, the variation in the reactor coolant temperature is maximized following the reactor trip at full power (FP) and the water volume is decreased in RCS. Accorgingly, pressurizer water level reaches the lowest. Thus, the 1st, 3rd and 5th transients mentioned previously are selected to analyze plant performance under RCS cooldown transients.

For the RCS heatup transients, 10%FP step load reduction transient and complete loss of load transient were analyzed. Nominal average coolant temperature and pressurizer water level become the highest value under full power operation conditions. Turbine load reduction causes a sudden reduction in steam flow and the heat transfer rate in the steam generator is reduced, causing the reactor coolant temperature to rise, which in turn causes coolant expansion, pressurizer insurge and RCS pressure rise. Thus, full power operation is selected as an initial condition for 2nd to 4th RCS heatup transients mentioned previously.

TABLE 1 MAIN INITIAL CONDITIONS AND ASSUMPTIONS

| Parameters | Value |
|---|---|
| 1. Initial Conditions[1]: | |
| Reactor Thermal Power (MWt) | 3400.0 |
| Average Coolant Temperature (°C) | 300.9 |
| Average Coolant Temperature at 15%FP / 30%FP (°C) | 293.0 /294.4 |
| RCS Pressure [MPa(g)] | 15.5 |
| Pressurizer Water Level (m) | 6.07 |
| 2. Pressurizer Design Parameters[2] : | |
| V (m[3]) / Din (m) / H (m) [3] | 59.5/ 2.54/ 12.63 |
| 3. Main Assumptions : | |
| 3.1 Control Systems Available | |
| Reactor Control System | Available |
| Rapid Power Reduction | Available |
| Steam Dump Control | Available |
| Feedwater Control (SG Level) | Available |
| Pressurizer Pressure Control | Available |
| Pressurizer water Level Control | Available |
| 3.2 Valve Effectiveness | |
| Pressurizer Safety Valves | Available |
| Secondary Steam Relief Valves | Unavailable |
| Secondary Steam Safety Valves | Available |

Note: 1. Nominal value at full power is provided except indication for all transients.
2. V: Internal Free Volume; Din: Inside Diameter; H: Internal Height.
3. Internal bottom is taken as reference elevation. Nominal water level of 45.0%H and high water level trip setpoint is 71%H.

### *Analysis Methods and Assumptions*

The operational transient analysis has been performed with RELAP5 thermal -hydraulics computer code to analyze the system behavior and plant control system performance for AP1000 nuclear power plant (NPP).

In the analysis, primary and secondary systems necessary for accident analysis are simulated such as

reactor and core with point kinetics model, RCS primary system (including reactor coolant pump, pressurizer and steam generator), the secondary system (including turbine, main feedwater system and related components), pressurizer safety valves, secondary steam relief valves and safety valves, and safety related systems. In addition, normally available support systems (i.e. plant control systems) are also simulated for the operational transient analysis such as reactor control system, rapid power reduction system, steam dump control, feedwater control (SG level control), pressurizer pressure control (i.e. pressurizer heater and spray), and pressurizer water level control systems etc[2][4]. An AP1000 plant nodalization diagram is shown in Figure 1.

In AP1000 plant operational transient analysis, the main initial conditions and analysis assumptions are listed in Table 1.

## Analysis Results

### *Step Load Increase from 15%FP to 25%FP*

A sudden increase in steam flow will cause a decrease in RCS temperature and cause a power mismatch between the reactor power and the steam generator load demand. And it causes an increase in reactor power by the actuation of the reactor regulation system via the control rod withdrawal until the plant reaches a new equilibrium condition at a higher power level corresponding to the increase in steam flow.

In the analysis, the reactor control system is assumed to be available and no credit is taken for the pressurizer pressure control and water level control.

As listed in Table 2, it's shown that pressurizer low pressure trip will not be actuated. Figure 2 to Figure 4 provides reactor power, average coolant temperature, pressurizer pressure and water level as a function of transient time.

### *Step Load Decrease from 100%FP to 90%FP*

A sudden decrease in steam flow will cause an increase in RCS temperature and cause a power mismatch between the reactor power and the steam generator load demand. And it causes a decrease in reactor power by the actuation of the reactor regulation system via the control rod insertion until the plant reaches a new equilibrium condition at a lower power level corresponding to the decrease in steam flow.

In this analysis, reactor control system is assumed to be available and no credit is taken for the pressurizer pressure control and water level control.

As listed in Table 2, it's shown that pressurizer high pressure trip will not be actuated. Figure 5 to Figure 7 provides reactor power, average coolant temperature, pressurizer pressure and water level vs transient time.

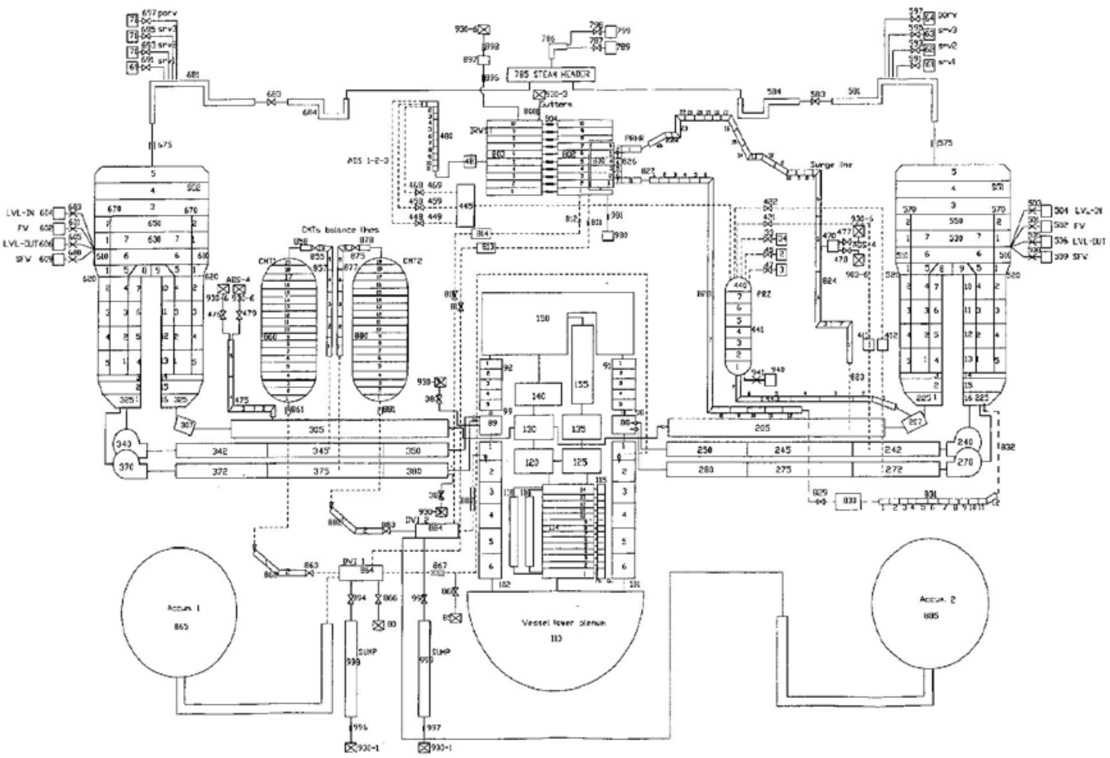

FIG. 1 AP1000 PLANT NODALIZATION DIAGRAM

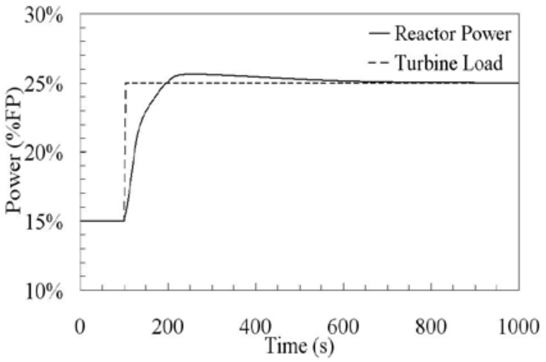


FIG. 2 REACTOR POWER & TURBINE LOAD TRANSIENTS UNDER STEP LOAD INCREASE FROM 15%FP TO 25%FP

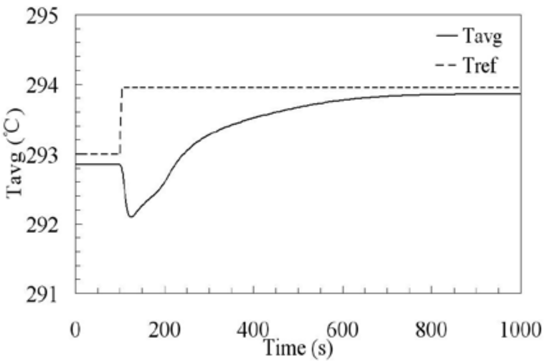


FIG. 3 AVERAGE COOLANT TEMPERATURE TRANSIENT UNDER STEP LOAD INCREASE FROM 15%FP TO 25%FP

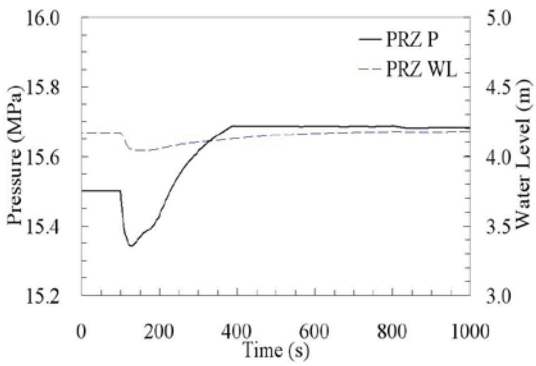


FIG. 4 PRESSURIZER PRESSURE & WATER LEVEL TRANSIENTS UNDER STEP LOAD INCREASE FROM 15%FP TO 25%FP

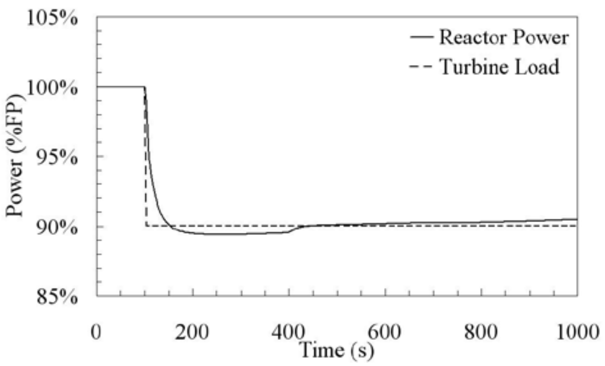


FIG. 5 REACTOR POWER & TURBINE LOAD TRANSIENTS UNDER STEP LOAD DECREASE FROM 100%FP TO 90%FP

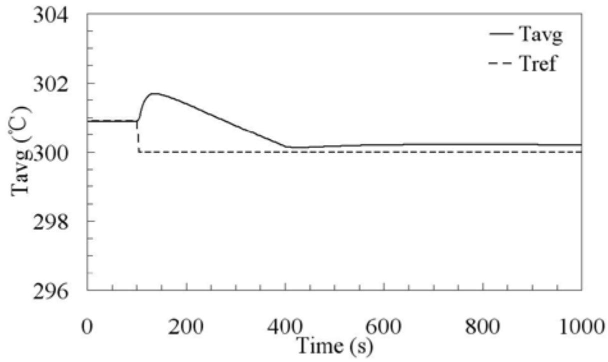


FIG. 6 AVERAGE COOLANT TEMPERATURE TRANSIENT UNDER STEP LOAD DECREASE FROM 100%FP TO 90%FP

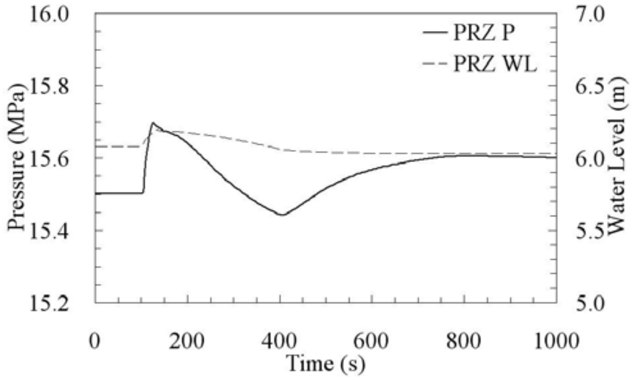


FIG. 7 PRESSURIZER PRESSURE & WATER LEVEL TRANSIENTS UNDER STEP LOAD DECREASE FROM 100%FP TO 90%FP

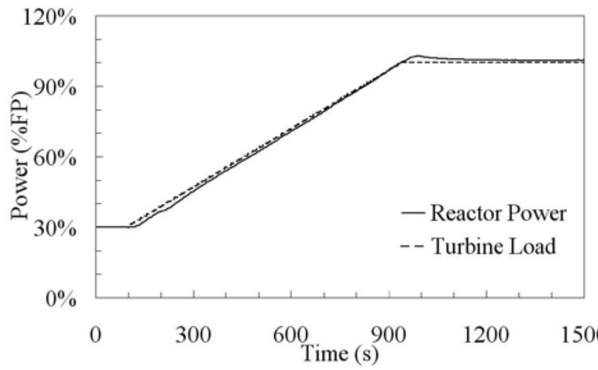


FIG. 8 REACTOR POWER & TURBINE LOAD TRANSIENTS UNDER RAMP LOAD INCREASE FROM 30% TO 100%FP

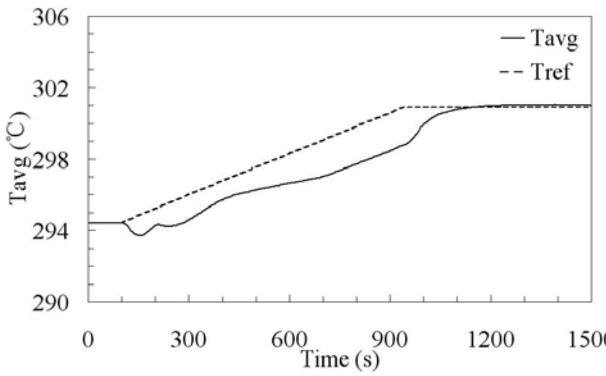


FIG. 9 AVERAGE COOLANT TEMPERATURE TRANSIENT UNDER RAMP LOAD INCREASE FROM 30% TO 100%FP

TABLE 2 SELECTED TRANSIENT ANALYSIS RESULTS FOR AP1000 PLANT

| Selected Operational Transients | Analysis Results | | Note |
|---|---|---|---|
| | Parameters[1] | Value | |
| Step Load Increase from 15%FP to 25%FP | PRZ Lowest WL (m) | 4.04 | No PRZ heater's uncovery[2] |
| | PRZ Max. P (MPa) | 15.69 | PRZ SV is not actuated[3] |
| | PRZ Min. P (MPa) | 15.34 | |
| | PRZ SV Status | No Actuation | |
| | Reactor Power Overshoot | <3%FP | |
| Step Load Decrease from 100%FP to 90%FP | PRZ Highest WL (m) | 6.19 | PRZ high WL trip inactuated[4] |
| | PRZ Max. P (MPa) | 15.70 | PRZ SV is not actuated[3] |
| | PRZ SV Status | No Actuation | |
| | Reactor Power Overshoot | <3%FP | |
| Ramp Load Increase from 30%FP to 100%FP | PRZ Highest WL (m) | 6.40 | PRZ high WL trip inactuated[4] |
| | PRZ Max. P (MPa) | 15.64 | PRZ high P trip inactuated[4]<br>PRZ SV is not actuated[3] |
| | PRZ Min. P (MPa) | 15.35 | PRZ low P trip inactuated[4] |
| | PRZ SV Status | No Actuation | |
| | Reactor Power Overshoot | <3%FP | |
| Load Rejection to House Load at Full Power | PRZ Highest WL (m) | 6.20 | PRZ high WL trip inactuated[4] |
| | PRZ Max. P (MPa) | 15.70 | PRZ SV is not actuated[3] |
| | PRZ SV Status | No Actuation | |
| | MSSV Status | No Actuation | |
| | Turbine Power | 5%FP | No turbine trip (House Load) |
| Reactor Trip & Turbine Trip at Full Power | PRZ Lowest WL (m) | 4.57 | No PRZ heater's uncovery[2]<br>No CMT Actuation[5] |
| | PRZ Min. P (MPa) | 13.65 | PRZ Low P SI inactuated[5] |

Note:

1. PRZ: Pressurizer, WL: Water Level, P: Pressure, Max.: Maximum, Min.: Minimum, SV: Safety Valve, PORV: Power Operated Relief Valve, MSSV-Main Steam Safety Valve, SI: Safety Injection, NTS: Nominal Trip Setpoint.
2. Pressurizer heater's height: 1.56 m.
3. Pressurizer SV openning setpoint: 17.24 MPa.
4. PRZ related reactor trip setpoints: High P Trip: **16.57**(16.79) MPa, Low P Trip: **13.52**(13.30) MPa, [**Analysis value** (NTS)], High WL Trip: **7.67** m [**59.8 %**( 71%)].
5. PRZ related ESF actuation setpoints: Low P SI Actuation: **12.83**(12.61) MPa, [Analysis value (NTS)], Low WL CMT Actuation: **3.52** m [**21.2%**(10%)].
6. Pressurizer Internal Height: 12.63 m, internal bottom is taken as reference elevation (WL=0.0 m).
7. Secondary Valves: PORV and MSSV openning setpoints are 7.88 MPa and 8.27 MPa respectively.

### *+5%/min Ramp Load Increase from 30% to 100%FP*

A gradual increase in steam flow will cause a decrease in RCS temperature and cause a power mismatch between the reactor power and the steam generator load demand. And it causes an increase in reactor power by the actuation of the reactor regulation system via the control rod withdrawal until the plant reaches a new equilibrium condition at a higher power level corresponding to the increase in steam flow.

In the analysis, plant control systems are assumed to be available.

As listed in Table 2, it's shown that pressurizer high pressure and low pressure trips will not be actuated. Figure 8 to Figure 10 provides reactor power, average coolant temperature, pressurizer pressure and water level a function of transient time.

### *Load Rejection to House Load at Full Power*

Load rejection will cause a sudden reduction in steam flow to house load, resulting in an increase in pressure and temperature in the steam generator. As a result, the heat transfer rate in the steam generator is reduced, causing the reactor coolant temperature to rise, which in turn causes coolant expansion, pressurizer insurge, and RCS pressure rise.

In the analysis, plant control systems are assumed to be available.

Figure 11 to Figure 15 provides reactor power, average coolant temperature, pressurizer pressure and water level, steam dump flow and secondary pressure vs. transient time.

As the rapid power reduction system (RPR) is designed to be able to rapidly reduce nuclear power by around 50%, so as to reach a power level that can be handled by steam dump system (SDCS) and reactor control system (RECS). The steam dump control system is to provide an artificial steam load with a capacity of 40% rated steam flow during rapid large load reductions (e.g.

turbine trip, large load rejection). Thus, the plant control systems (including RPR, SDCS and RECS) will be actuated under load rejection to house load at full power.

As shown in Table 2, reactor trip and turbine trip will not be actuated, as well as pressurizer safety valve and main steam safety valves. Plant is able to continue stable operation with the minimal house load.

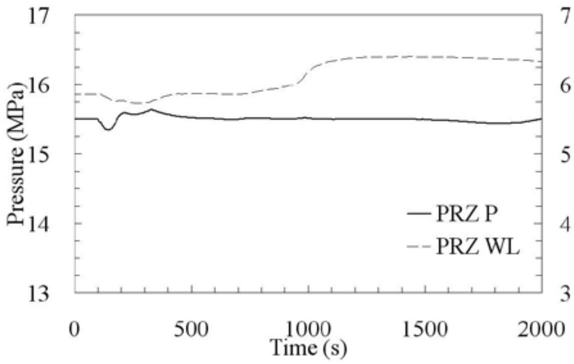


FIG. 10 PRESSURIZER PRESSURE & WATER LEVEL TRANSIENTS UNDER RAMP LOAD INCREASE FROM 30% TO 100%FP

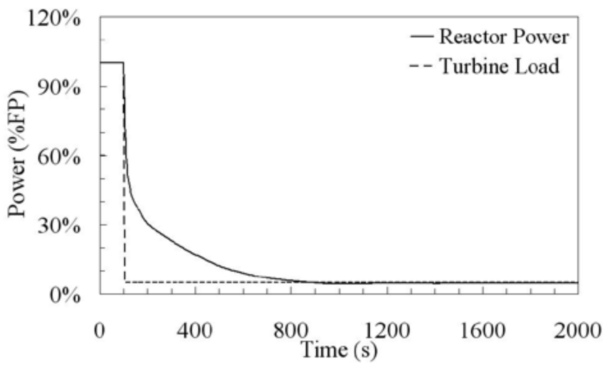


FIG. 11 REACTOR POWER & TURBINE LOAD TRANSIENTS UNDER LOAD REJECTION AT FULL POWER

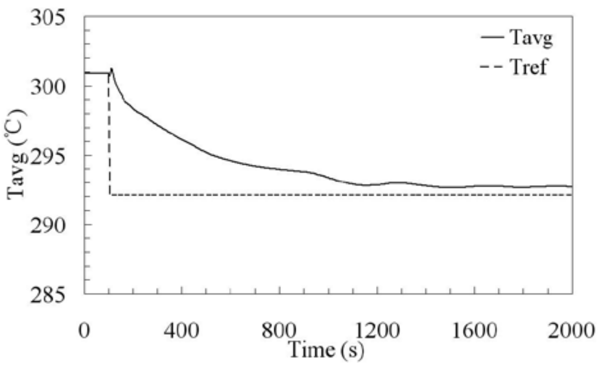


FIG. 12 AVERAGE COOLANT TEMPERATURE TRANSIENT UNDER LOAD REJECTION AT FULL POWER

### *Normal Reactor Trip at Full Power*

Reactor trip will cause a sudden reduction in reactor power, resulting in a decrease in pressure and temperature in RCS. Following the reactor trip, the turbine will be tripped, causing the reactor coolant temperature decrease, which in turn causes coolant shrinkage, pressurizer outsurge, and RCS pressure reduction.

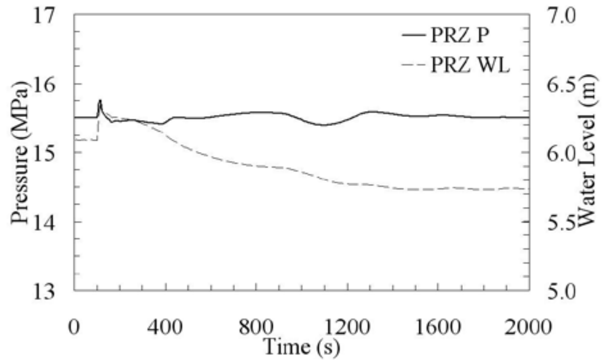


FIG. 13 PRESSURIZER PRESSURE & WATER LEVEL TRANSIENTS UNDER LOAD REJECTION AT FULL POWER

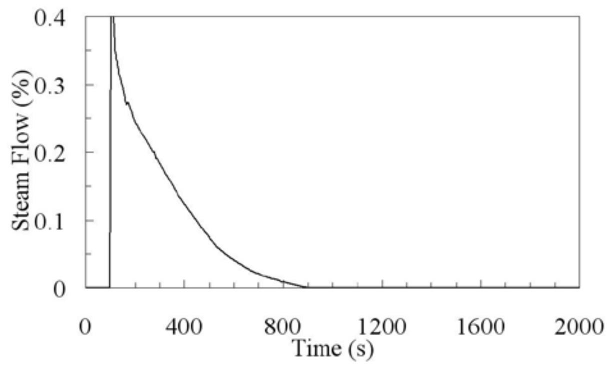


FIG. 14 STEAM DUMP FLOW TRANSIENT UNDER LOAD REJECTION AT FULL POWER

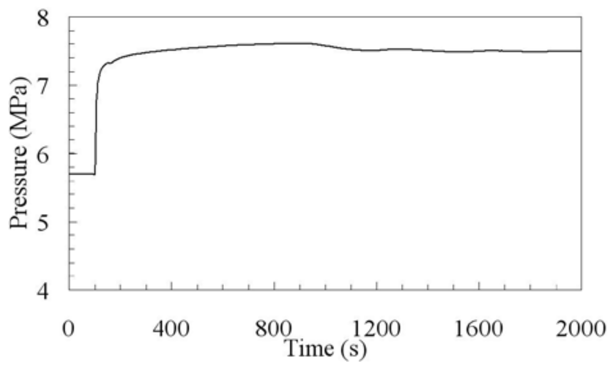


FIG. 15 SECONDARY PRESSURE TRANSIENT UNDER LOAD REJECTION AT FULL POWER

In the analysis, plant control systems are assumed to be available.

As listed in Table 2, it's shown that the pressurizer heater will not be uncovered while the pressurizer low pressure safety injection signal is not actuated. There is an appropriate margin between the minimum pressurizer pressure (13.65 MPa) and the low pressure

safety injection actuation setpoint (12.83 MPa). Figure 16 to Figure 19 provides reactor power, average coolant temperature, pressurizer pressure and water level, and steam dump flow vs. transient time.

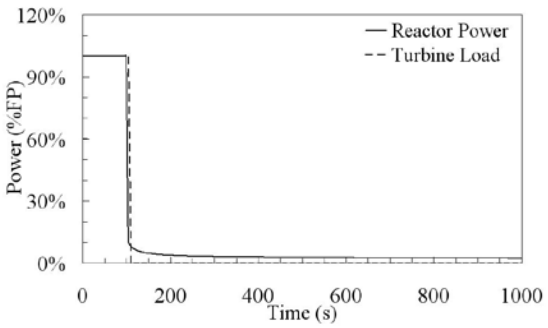


FIG. 16 REACTOR POWER & TURBINE LOAD TRANSIENTS UNDER NORMAL REACTOR TRIP AT FULL POWER

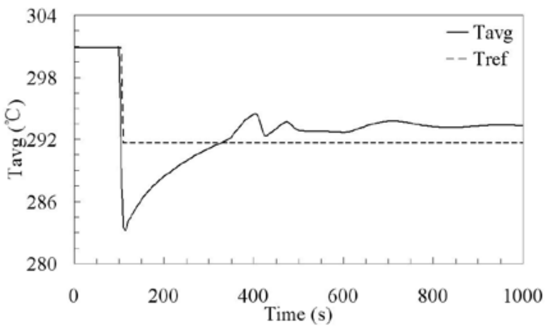


FIG. 17 AVERAGE COOLANT TEMPERATURE TRANSIENT UNDER NORMAL REACTOR TRIP AT FULL POWER

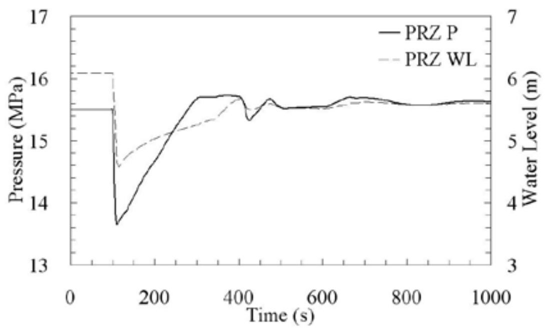


FIG. 18 PRESSURIZER PRESSURE & WATER LEVEL TRANSIENTS UNDER NORMAL REACTOR TRIP AT FULL POWER

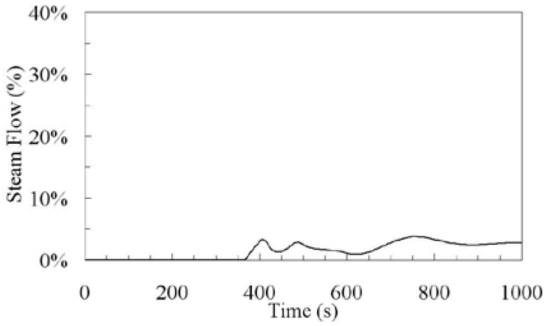


FIG. 19 STEAM DUMP FLOW TRANSIENT UNDER NORMAL REACTOR TRIP AT FULL POWER

## Summary

For AP1000 plant design, the operational transient analysis has been performed with RELAP5 thermal-hydraulic computer code under five typical operational transients i.e. (a) step load increase from 15% FP to 25%FP, (b) step load decrease from 100%FP to 90%FP, (c) +5%/min ramp load increase from 30%FP to 100%FP, (d) load rejection to house load at full power, (e) normal reactor trip at full power.

TABLE 3 ACCEPTANCE CRITERIA

| Acceptance Criterion | Case No.[1, 2] | | | | |
|---|---|---|---|---|---|
| | 1 | 2 | 3 | 4 | 5 |
| No reactor trip | √ | √ | √ | √ | NA |
| No turbine trip | √ | √ | √ | √ | NA |
| Pressurizer safety valve remains closed | √ | √ | √ | √ | √ |
| Pressurizer heaters remain covered | √ | √ | √ | √ | √ |
| No secondary valve actuation[3] | √ | √ | √ | √ | √ |
| PORV remains closed | √ | √ | √ | √ | √ |
| MSSV remains closed | √ | √ | √ | √ | √ |
| No nuclear power overshoot by more than 3% | √ | √ | √ | √ | NA |
| No high frequency rod stepping | √ | √ | √ | √ | NA |
| No ESF actuation[4] | √ | √ | √ | √ | √ |
| No CMT/ADS Actuation | √ | √ | √ | √ | √ |

Note: 1. Case definition :
Case 1-Step load increase from 15% to 25%FP, Case 2-Step load decrease from 100% to 90%FP, Case 3-Ramp load change from 30% to 100%FP, Case 4-Load rejection to house load at 100%FP,
Case 5-Normal reactor trip.
2. Symbol indication: √-Acceptance criteria satisfied, NA-Not applicable.
3. ESF-Engineered safety features,
PORV-Main steam relief valve,
MSSV- Main steam safety valve.
4. No ESF setpoints is challenged.

The preliminary operational transient analysis results indicate that the design requirements are satisfied under five selected operational transients. There is an appropriate margin between the operation parameter and specified setpoint for the reactor trip, ESF actuation, primary and secondary valves actuation under the selected operational transients.

AP1000 NSSS control systems are able to achieve acceptable performance for the ±10% step load change, 5%/min ramp load change, load rejection to house load and normal reactor trip transients. Acceptability of the AP1000 plant control system against the acceptance criteria is provided in Table 3.

Based on the above information, it's concluded that the specified acceptance criterias are satisfied under the selected operational transients. AP1000 NSSS control

systems could control the plant operation with acceptable performance under the specified operational transients.

For AP1000 plant design, those requirements of ALWR URD could be satisfied for AP1000 NSSS control systems.

## REFERENCES

"Advanced Light Water Reactor (ALWR) Utility Requirement Document (URD)", Volume II, ALWR Evolutionary Plant, US EPRI, CA, USA, December 1995.

"AP1000 Plant Description and Safety Analysis Report", WCAP-15612 (Non- Proprietary), US Westinghouse Electric Co. LLC, PA, USA, Dec. 2000.

"A1000 Design Control Documents (DCD) Chapter 3, Chapter 7, Chapter 16", Rev.19, US Westinghouse Electric Co.LLC, PA, USA, 2011.

ZHENG Limin, "Pressurizer Volume Demonstration Analysis", ICONE13-50569, Proceeding of 13th International Conference on Nuclear Engineering (ICONE-13), Beijing, China, May 2005.